

2.0 SITE HISTORICAL OVERVIEW

2.1 BACKGROUND

Established in 1942, the Oak Ridge Reservation (ORR) occupies approximately 34,500 acres within the city boundaries of Oak Ridge, Tennessee. Of the three major DOE facilities on the ORR, the Y-12 Complex serves as the primary location for Defense Program missions. The Y-12 Complex industrial plant occupies approximately 3,400 acres, with a surrounding buffer zone of an additional 2,800 acres (Fig. 2.1-1). The plant is situated in Bear Creek Valley near the eastern boundary of the ORR, approximately three miles from the population center of the city of Oak Ridge. The plant site is bounded on the south by Chestnut Ridge and on the north by Pine Ridge. This site was originally chosen for the Electromagnetic Plant, which initially occupied 825 acres. The Electromagnetic Plant used staged calutrons (production mass spectrographs) to produce enriched uranium for the Manhattan Project.



Fig. 2.1-1 The Y-12 Complex.

After the electromagnetic uranium enrichment process was rendered obsolete by the gaseous diffusion process in the mid-1940s, the Y-12 Complex became an enriched uranium weapons component production facility. Since then, the Y-12 Complex has become a center for handling, processing, manufacturing, assembling, storing, and disassembling uranium material and nuclear weapons components. Material processing has included the recovery of highly enriched recycled uranium (RU) from reactor returns. Today, the Y-12 Complex's mission primarily consists of dismantling nuclear weapons components and serving as DOE's primary repository for highly enriched uranium (HEU).

2.2 CHARACTERIZATION OF RU STREAMS RECEIVED AT THE Y-12 COMPLEX

Uranium streams received at the Y-12 Complex that contained or may have contained RU constituents included:

- highly enriched RU material in the form of uranyl nitrate solutions or uranium oxide received from Savannah River and ICPP and
- slightly depleted RU oxide (including ash and scrap) from ORGDP, Hanford, and PGDP.

In accordance with the methodology prescribed by the DOE Project Plan,¹ calculations were performed to estimate for these streams the additional dose presented by constituents in irradiated uranium over that of unirradiated uranium. A fractional dose calculation with a result of <0.1 indicates that the additional dose presented by the RU constituents is less than 10% of the dose expected from doing similar work with uncontaminated weapons-grade uranium. RU streams characterized by a dose fraction of <0.1 were deemed *de minimis* in accordance with the definition established by DOE for the Recycled Uranium Mass Balance Project. For those streams, the radiation-protection measures in place for the presence of uranium are considered adequate for worker protection (see Appendix A).

The highly enriched RU from Savannah River and Idaho in the form of uranyl nitrate and uranium oxide was processed at the Y-12 Complex and shipped to Savannah River as HEU metal for fabrication of production reactor fuel. The primary focus of this document is on the facilities and processes that had the potential for concentrating the RU constituents, relative to the uranium flow, and so presented the greatest potential for increased worker exposure.

Slightly depleted RU oxide was received by the Y-12 Complex from ORGDP, Hanford, and PGDP (including fluorination tower ash from PGDP). Documentation and discussion with many individuals who worked at the Y-12 Complex from the 1950s onward indicated that the plant did not have the need for nor the capability of chemically processing this material. Therefore, it is assumed this material was sent to the plant for storage prior to burial or further disposition to other Oak Ridge Operations sites; most of the ash was returned to PDGP. Since these materials were apparently not processed or handled directly at the Y-12 Complex, they are not at this time considered to be potential sources of increased personnel exposure or significant environmental release. Further analysis may be warranted in the future if these materials are determined to have been processed at the Y-12 Complex.

Depleted uranium metal from Fernald, produced from gaseous diffusion plant tails, has been used extensively in weapons and defense programs at Idaho, Rocky Flats, the Y-12 Complex, and other sites. Identical material received at Idaho was analyzed in the Report on Mass Balance at the Specific Manufacturing Capability Project² where it was determined that the fractional dose resulting from the RU constituents is less than 10% of that of the uranium itself. The ORGDP Mass Balance Report also confirms very low levels of transuranics and Tc in the tails streams. Processing of this material in a manner that concentrated the RU constituents was not performed at the Y-12 Complex; rather, the material was fabricated as is into an end-use form. For this reason, and in accordance with the DOE Project Plan, this depleted uranium metal stream was excluded from further consideration.

¹ U.S. Department of Energy, *Historical Generation and Flow of Recycled Uranium in the DOE Complex*, Appendix A, February 2000.

² Barg, Don C., *TRU and DU at SMC, Report on Mass Balance at SMC*, June 19, 2000.

2.3 KEY URANIUM-PROCESSING FACILITIES AT THE Y-12 COMPLEX

Six locations within the Y-12 Complex were involved in the highly enriched RU-processing operations (Fig. 2.3-1). Until the early 1970s, chemical processing of highly enriched RU occurred in the large 9212 complex. Afterward, chemical processing, following virtually the same procedures and using nearly identical equipment, occurred in Building 9206. Enriched uranium product was stored in Building 9720-5. The S-3 Ponds served as impoundment for process wastewater until the mid-1980s; the four earthen basins comprising the S-3 Ponds had no direct discharge to any local creek or river tributary. After the ponds were closed, RU-process wastewater was treated by a variety of methods at the West End Treatment Facility (WETF). Treated wastewater was discharged from WETF to East Fork Poplar Creek. New Hope Pond served as a surface-water impoundment that captured entrained solids from rainwater and secondary wastewaters.

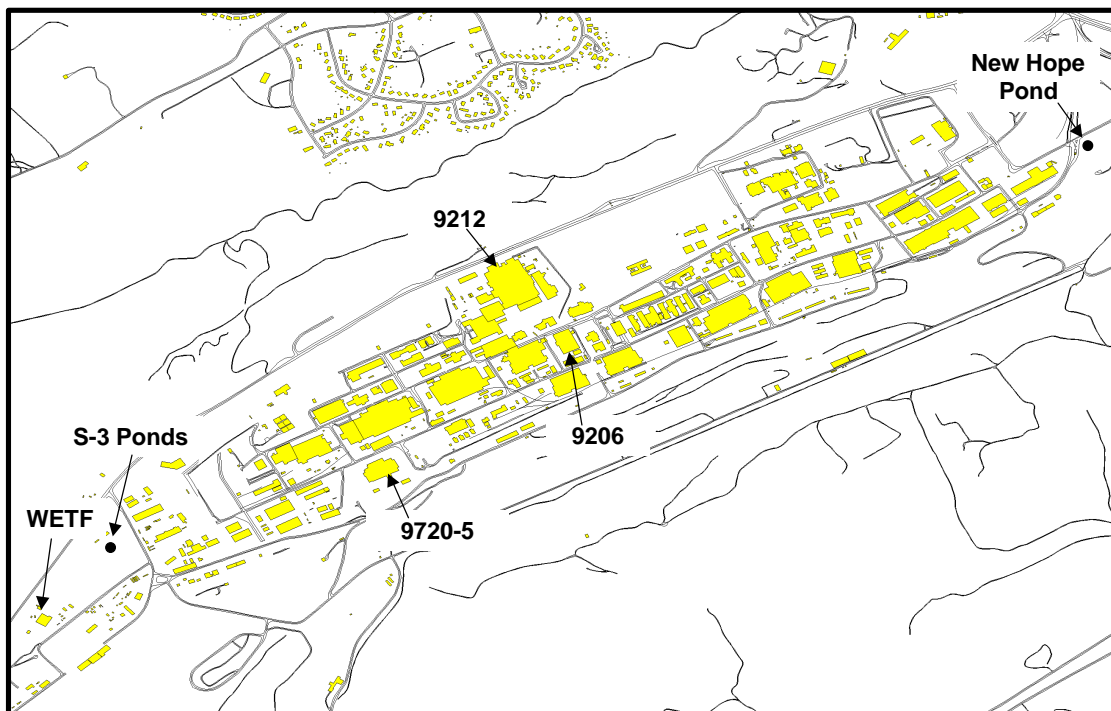


Fig. 2.3-1 RU Operations Occurred in Six Facilities at the Y-12 Complex.

Building 9212 Complex

The 9212 Complex processes HEU to produce uranium metal and oxide suitable for storage, reactor fuels, specialty compounds, or weapons components. The recovery and purification operations extract HEU from uranium-bearing scrap and waste and process it into forms suitable for reuse or accountability. The majority of this scrap and waste is generated by the Y-12 Complex's weapons production or disassembly operations and by

the recovery processes themselves. Some scrap and waste is generated through nuclear materials production; additional scrap is received from other sites for recovery or for accountability of the uranium it contains. The nature of these uranium-bearing materials varies from combustible and noncombustible solids to aqueous and organic solutions. Concentrations of uranium vary in these materials from pure uranium compounds and alloys to trace quantities [parts per million (ppm) levels] in combustibles and solutions.

The 9212 Complex includes Buildings 9212, 9809, 9812, 9818, 9815, and 9980. Over 100 operations or processes have been, or are capable of being, performed within this complex.

The largest building, 9212 was constructed in the early 1940s. The building is a multistory facility constructed of structural steel frame infilled at the perimeter with thick hollow clay tile. The substructure basement is constructed of reinforced concrete. The original structure consisted of a central building (the “Headhouse”) 72 feet wide by 308 feet long (N-S direction) and four parallel wings projecting from the east side of the Headhouse, each 36 feet by 264 feet (A, B, C, and D Wings); open space between the wings was designed to mitigate the impact of a postulated criticality accident or chemical explosion.

The original mission of Building 9212 during World War II was to recover HEU from the electromagnetic separation project. Recovery was accomplished in the four wings.

Following World War II, the 9212 building was expanded through a series of structural modifications and additions to accommodate the increased production of uranium from the Oak Ridge Gaseous Diffusion Plant (ORGDP) and to provide capability for the recovery of uranium from waste materials. In 1948, new structures were erected in the spaces between the existing A, B, C, and D Wings (these were called the A-1, B-1, and C-1 Wings) and adjoining D Wing (the D-1 Wing). Next, a single story 113-foot-wide by 400-foot-long steel frame structure was added in 1951 (the E Wing) adjacent to the D-1 Wing and north of the Headhouse. The E Wing was added to facilitate the casting and machining of uranium components. Other, less-extensive modifications and additions have subsequently been made.

In the late 1950s, continuous solvent extraction equipment was installed in the B-1 Wing and “penthouses” were raised on the roof to house 30-foot-long extraction columns. This period covered the transition from small-scale batch operations to the existing continuous recovery equipment in use today.

The uranium hexafluoride conversion facility in the D Wing was shutdown in 1964, essentially halting the introduction of new HEU metal into the weapons stockpile. Since 1964, all HEU weapons components have been produced with uranium recovered from retired weapon subassemblies and production scrap. Special projects, such as the production of fuel for the NASA Rover Project and various research reactors, were accomplished in Building 9212 from time to time.

A number of facility modifications have been performed to reduce the environmental impact of the operations. These modifications were both in response to changing regulations as well as an effort to maintain exposures to ionizing radiation as low as reasonably achievable (ALARA).

Building 9212 currently performs four primary functions:

- casting of HEU metal (for weapons, reactors, storage, or other uses),
- accountability of HEU from plant activities (quality evaluation, casting, storage),
- recovery of HEU to a form suitable for storage (from plant activities and commercial scrap), and
- serving as the U.S. source of all HEU used in test, research, or propulsion reactors and for isotope production.

In addition to these primary missions, Building 9212 supports International Atomic Energy Agency (IAEA) sampling of surplus enriched uranium, packaging HEU for off-site shipment, and producing specialized uranium compounds and metal for research reactor fuel.

The recovery and purification process for HEU relies on the unique physical and chemical properties of uranium in a nitric acid system, where uranium forms uranyl nitrate [$\text{UO}_2(\text{NO}_3)_2$], abbreviated UN; when concentrated to the point of crystallization, the nitrate becomes uranyl nitrate hexahydrate [$\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$], abbreviated UNH. The approach to recovery and purification, therefore, consists of chemically changing HEU into a nitrate solution through dissolution, leaching, and other processes and using the chemical properties of uranium to concentrate, purify, extract, and finally convert the HEU into a purified metallic form. The recovery process generally includes the following steps:

1. “Headend” (first-step) operations (Headhouse; B-1, C, and C-1 Wings)
 - bulk volume reduction of scrap (mostly burning)
 - dissolution of scrap into uranyl nitrate solution
 - separation of uranium from non-uranium materials
2. Continuous purification and chemical conversion operations (B-1, C-1, and C-Wings)
 - organic solvent extraction
 - evaporation
 - conversion of uranyl nitrate to UO_3
 - conversion of UO_3 to uranium tetrafluoride (UF_4 or greensalt)
3. Reduction (E-Wing)
 - blending of UF_4
 - calcium reduction of UF_4 powder to uranium metal
4. Special processing (E-Wing)
 - special materials production
 - accountability of scrap
 - scrap dissolution
 - packaging of HEU materials for shipment

5. Waste streams and materials recovery (Buildings 9212, 9809, 9812, 9818, and 9815)
- nitrate recycle
 - biodenitrification
 - materials storage and handling
 - chemical make-up
 - organic handling
 - neptunium recovery

Building 9206 Complex

Building 9206 is centrally located in the Y-12 Complex below Building 9212. Approximately 260 feet long and 165 feet wide, this building is a multistory facility constructed in the early 1940s of structural steel infilled with thick hollow clay tile at the perimeter. It has a 43,614-ft² first story with a 19,800-ft² second story in its central portion, a 3,300-ft² mezzanine, and a 580-ft² penthouse. The 9206 building has been used extensively over its lifetime for the chemical processing of uranium.

Building 9206 has several related structures that house supporting or process services and/or equipment, all of which are considered inclusively as the 9206 Facility. These are 9768, 9720-17, 9409-17, 9510-2, 9767-2, and the east and west tank farm pits.

Enriched uranium processes, activities, and/or missions of the 9206 Facility have included:

- chemical recycle, charge preparation, HEU recovery, and product processing for the electromagnetic process (1945 to 1946);
- recovery of enriched uranium [both HEU and low-enriched uranium (LEU)] from Y-12 Complex programs and many other sites (1947 to 1994);
- production of uranium compounds for other sites (1949 to 1972);
- conversion of UF₆ to UF₄ to uranium metal for weapons (1954 to 1964);
- casting and machining of HEU metal for weapons (1955 to 1965);
- recovery of HEU from Savannah River Site (SRS) solutions and other scrap for return to SRS as uranium metal (1972 to 1989);
- conversion of excess HEU metal to oxide feed for the Portsmouth Gaseous Diffusion Plant (1980 to 1985); and
- storage of in-process materials (1950 to present).

Non-enriched uranium processes, activities, and/or missions of the 9206 facility have included:

- recycling depleted uranium chips (1951 to late 1950s),
- production of uranium compounds for other sites (1949 to 1972),
- canning of normal-assay uranium slugs for nuclear reactor use (1950 to 1952), and
- storage of in-process materials (1950 to present).

Non-uranium processes, activities, and/or missions of the 9206 facility have included:

- zirconium processing (1950),
- thorium parts processing (1963),
- graphite flour processing and preparation of special organic compounds [isotuxene (ITX), cinnamylideneindene (CAI), and pitch] in support of the Rover Program (1967 to 1971), and
- radiogenic lead processing (1965 to 1966).

Building 9206 is currently used for in-process materials storage. This will continue to be the function of 9206 until the stored material can be transferred to Building 9212 for processing or transferred to another storage location.

Other Uranium Handling Facilities

Building 9720-5

Used as a warehouse for short- and long-term storage of strategic materials, Building 9720-5 was built in 1944 and has been renovated several times. The facility is a single-story building located in the southwestern portion of the Y-12 Complex. It has a concrete floor elevated about 1 meter above the local grade and five dock areas; air is exhausted unfiltered through roof-mounted fans. The main warehouse dimensions are approximately 150 ft x 300 ft. Building 9720-5 is a shipping/receiving facility for special nuclear material (SNM) and the primary storage facility for interim and prolonged low-maintenance storage of HEU.

S-3 Ponds

The S-3 Ponds consisted of four unlined earthen basins constructed at the west end of the plant to receive acid wastewater from various Y-12 Complex production operations involving both enriched and depleted uranium streams. These basins were placed into operation around 1951 and were taken out of service in 1984. Various metal impurities and radionuclides stripped from HEU in the 9212 and 9206 solvent extraction steps (approximately 10% to 30% of the RU, Pu, Np, and Tc) were discharged with the dilute nitric acid and other process-derived acid wastewater to the S-3 Ponds prior to the mid-1980s. Uranium-containing process wastewaters from various depleted uranium plant operations were also discharged to the S-3 surface impoundment. The ponds were closed and capped in the mid-1980s.

West End Treatment Facility

Beginning in the mid-1980s, after discontinuance of the use of the S-3 Ponds, the West End Treatment Facility (WETF) was constructed for treating mixed low-level waste (LLW) and LLW-contaminated wastewater generated by Y-12 Complex production and other DOE ORO processes meeting the facility waste acceptance criteria and Resource

Conservation Recovery Act (RCRA) Permit-by-Rule regulations. Nitrate wastewater contaminated with enriched uranium (EU) was mixed with much larger quantities of wastewater contaminated with depleted uranium. Consequently, the EU component was diluted to less-than-normal-assay uranium. Treatment methods include hydroxide precipitation of metals, sludge settling and decanting, biodenitrification, bio-oxidation, pH adjustment, degasification, coagulation, flocculation, clarification, filtration, and carbon absorption. Wastewaters are discharged from the facility under NPDES permit into East Fork Poplar Creek. Contaminated sludges generated by the WETF operations are pumped into one of four large (0.5-million gallon) sludge storage tanks.

New Hope Pond

In the 1950s, New Hope Pond was constructed and placed in operation to provide a holdup basin on East Fork Poplar Creek at the east end of the Y-12 Complex. The pond facilitated mixing and offered a sampling point for rainwater runoff, once-through cooling water, steam plant boiler blow-down, and secondary production process wastewaters. New Hope Pond was also used as a settling basin to remove entrained coal fines from the Y-12 Complex coal yard. At the same time, the pond functioned to remove any suspended contamination from rainwater, miscellaneous releases from various tank farms and storage yards, and inadvertent releases from process buildings. In 1973, New Hope Pond was dredged, and the sludge was transferred to a basin located on Chestnut Ridge; this process was repeated in the latter 1980s as part of an environmental restoration project. Data from a leach test showed that the sediment was not hazardous (see Section 4.5.2.3).

2.4 Y-12 COMPLEX OPERATIONS INVOLVING RU

The RU streams at the Y-12 Complex encompassed a variety of material forms, including uranyl nitrate solutions, molten UNH, UO_3 , UO_2 , UF_4 , uranium metal, uranium alloys, and a variety of associated wastes. These RU streams impacted a number of plant facilities. Those with significant involvement with RU were Buildings 9212, 9206, 9720-5, the S-3 Ponds, and the West End Treatment Facility. New Hope Pond experienced less impact. With the exception of the S-3 Ponds, which have been closed in place and capped, and New Hope Pond, which has been closed after draining and removing the sediment, all of these facilities continue to be used today.

From 1953 until the early 1970s, all processing of SRS and the Idaho Chemical Processing Plant (ICPP) RU material to metal product was performed in Building 9212. From the early 1970s until 1989, most activities involving processing RU material to metal product were performed in Building 9206. In Building 9212, however, there continued to be evaporation and concentration of RU-derived uranyl nitrate solutions before transfer to 9206 and also sampling, fracturing, and packaging of RU-derived metal product prior to shipping.

Typically, SRS shipped uranyl nitrate solution to the Y-12 Complex in tanker trucks with capacities of 3,800–5,000 gallons. After primary evaporation, the material went through purification by solvent extraction, denitration to produce UO_3 , reduction to UO_2 ,

hydrofluorination to UF_4 , and “bomb” reduction to metal. After the metal was cleaned, it was prepared in 9212 for shipment back to SRS from 9720-5 or was stored. From 1972 to the early 1990s, SRS sent scrap from the uranium-aluminum (U-Al) alloy casting process to the Y-12 Complex for processing. This material was dissolved in sodium hydroxide (NaOH) solution to remove the aluminum and produce sodium diuranate solids. The sodium diuranate was dissolved in nitric acid to produce uranyl nitrate solution, which was then purified and converted to metal. The Y-12 Complex also processed furnace dross and floor sweepings from the SRS U-Al casting process. These materials were similarly processed in 9206 by dissolution, purification, and conversion to metal. However, not all of the U-Al material was processed, and some quantities remain in storage at the Y-12 Complex.

From 1953 until the late 1980s, ICPP processed spent Navy, research, and experimental reactor fuel to recover and recycle HEU. The resultant product was shipped to the Y-12 Complex for processing to metal and subsequent shipment to SRS (or storage). Initially, ICPP provided UN solution. However, after a denitrator was installed at ICPP in 1970, ICPP provided RU to the Y-12 Complex as UO_3 . After undergoing dissolution, the UO_3 was processed by the Y-12 Complex through the same steps as the uranyl nitrate solution.

2.5 CONCENTRATING PROCESSES

At the inception of the RU processing program at the Y-12 Complex, local radiation safety personnel developed strict limits on the allowable radioactivity that could enter the plant in RU. The plant RU acceptance criteria (see Section 4.3) were expressed in terms of activity ratios derived from allowable radiological limits for uranium, transuranic (TRU) elements, and fission products. As a direct result, control was achieved by limiting the quantities of TRU elements and various reactor fission products in relation to the associated uranium flows. This allowed existing uranium contamination control standards and practices for unirradiated HEU to be used for protection of plant workers from the incremental effects due to the presence of RU constituents. This radiation control philosophy presupposes that the RU constituents do not concentrate to any significant extent in the plant equipment or processes. In instances where significant concentration may occur, modified TRU limits may be required.

The objective of the Y-12 Complex RU work was recovery of HEU from various uranium scrap metals, oxides, and solutions for preparation of uranium metal for the DOE production reactors at SRS. While similar to chemical processing facilities used at Savannah River, Idaho, and Hanford to separate fission products and Pu from irradiated uranium fuel, the Y-12 Complex processes were designed and operated primarily to recover HEU from unirradiated production scrap and various process residues, remove problematic chemical impurities, such as iron, nickel, chromium, and carbon from the uranium stream, and convert the various uranium forms to uranium metal. The uranium processes were operated to minimize loss of HEU in the various waste streams. Consequently, incoming RU constituents other than uranium were left to distribute across the chemical facilities without any particular process control or design specific to RU.

The nature of the Y-12 Complex HEU processes is such that the RU constituents were not deliberately concentrated in any stream on an overall mass-to-volume basis. However, when considered on an unirradiated HEU basis, even the smallest RU stream, regardless of the absolute TRU or fission-product content, may become a stream where the TRU or fission products exist as the dominant isotopes whenever the uranium is selectively removed from the process stream. This situation occurred in both 9212 and 9206 operations during primary and secondary solvent extraction purification steps and, to a lesser extent, during acid leaching of certain process solids to recover the uranium. Overall, sizable fractions of the incoming radionuclides followed the uranium through the process and ultimately ended up in the HEU metal product shipped to Savannah River. However, in the waste stream, which was dilute in uranium by volume, TRU became concentrated with respect to uranium mass. The ^{236}U fed to the process in the RU partitioned with the uranium during all of the process steps because ^{236}U , for all practical purposes, is chemically and physically indistinguishable from ^{235}U , ^{234}U , and ^{238}U isotopes.

As an artifact of the chemical characteristics of TRU elements and fission products of concern in mixed aqueous-organic solutions (specifically, nitric acid-dibutyl carbitol), approximately 10% to 30% of the target radionuclides remained in the nitric acid feed stream after solvent extraction (raffinate). However, only a small fraction of the incoming uranium ended up in the primary solvent extraction raffinate stream, as intended. As a net result, even though less than half of the TRU elements and Tc ended up in the raffinate, these RU components were effectively concentrated in the primary waste discharge stream from the recovery operation. The secondary solvent extraction raffinate contained a significantly larger quantity of uranium, but the target radionuclides were still concentrated on a uranium basis, although to a lesser extent than the primary system. The raffinate from the secondary system was recycled back to the headend of the recovery process rather than being discharged.

RU constituents contained in the primary raffinate ultimately ended up in the S-3 Ponds or, after about 1985, in the WETF sludge tanks. Contaminated sludge was allowed to accumulate in the S-3 Ponds for more than 30 years before the ponds were taken out of service. The pond sludge was combined with a large quantity of depleted uranium from other plant operations. These other uranium streams did not contain significant RU. Hence, neither the S-3 Ponds or WETF created a significant RU concentration point (relative to uranium).

Other situations in which the RU constituents may become concentrated (relative to the uranium flow) occur when uranium is selectively removed from certain process-generated contaminated solids and during process-residue leaching operations, leaving a fraction of the TRU elements and fission products behind. The actual radiological hazard created by such operations is not particularly significant because any radionuclides left behind are fixed in the contaminated solids and relatively immobile.

2.6 ACTIVITIES WHERE WORKERS WERE LIKELY TO BE IN CONTACT WITH RU THROUGH DIRECT PHYSICAL CONTACT OR AIRBORNE DUST

In reviewing Y-12 Complex facilities and processes, the Project Team identified a number of activities that, based on available data and process knowledge, would be expected to present the greatest potential for workers to be exposed to the RU constituents of interest (i.e., ^{236}U , Pu, Np, and Tc). These activities are described in Table 2.6, which is subdivided by areas in which activities took place; the table includes information on time frame and occupational exposure potential (OEP) values. The potential for worker occupational exposure is expressed as High, Moderate, Low, or No Significant potential. These values have been qualitatively determined by the Project Team. To assign these values, the team reviewed activities and considered three parameters: the likelihood of material becoming airborne during the activity, the level of hazardous constituents in the airborne material, and the length of time a worker might be exposed to the airborne material. These were assigned numbers (0, 1, 2, or 3) and the product of the values for the three parameters determined the estimate of High, Moderate, Low, or No Significant (see Appendix B). Activities associated with long-term exposure to high levels of materials with high radiological activity received the highest rating, while short-duration activities in relatively “clean” areas received the lowest rating.

Table 2-6 Activities at the Y-12 Complex with Potential for Worker Exposure to RU

Location	Activity	Time Frame	Constituents	Occupational Exposure Potential*
1. Activities Associated with Building 9212				
9212	1A. ICPP UN solution received in safe bottles	1953-early 1970s	0.11 ppb Pu 4.7 ppm Np 0.13 ppm Tc 10% ^{236}U	No Significant
9929-1	1B. SRS tanker truck weighed for gross weight	1955-1988	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ^{236}U	No Significant
9212 Complex	1C. SRS material sampled	1953-1988	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ^{236}U	No Significant
9212	1D. ICPP UN solution poured into “pour-up” stations for transfer to intermediate storage tanks	1953-early 1970s	0.11 ppb Pu 4.7 ppm Np 0.13 ppm Tc 10% ^{236}U	Low
9212 Complex	1E. SRS UN solution pumped to 9212	1955-1988	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ^{236}U	No Significant
9212	1F. SRS and/or ICPP UN evaporated and concentrated	1953-1989	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ^{236}U	Moderate

Table 2-6 Activities at the Y-12 Complex with Potential for Worker Exposure to RU

Location	Activity	Time Frame	Constituents	Occupational Exposure Potential*
9212	1G. Manual filling and loading SRS and/or ICPP UN into safe bottles for transfer to 9206	1970s-1989	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1H. ICPP UO ₃ received, dissolved to form UN	1970s-1989	0.11 ppb Pu 4.7 ppm Np 0.13 ppm Tc 10% ²³⁶ U	Moderate
9212	1I. Purification of SRS and/or ICPP UN via solvent extraction (primary and secondary extraction)	1953-1970s	4.4 ppm Pu 5.9 ppm Np 190 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1J. Discard of solvent extraction raffinate to S-3 Ponds	1953-mid-1980s	3.5 ppm Pu 5.0 ppm Np 100 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1K. Feeding of raffinate to 9212 bioreactor	1970s-1989	3.5 ppm Pu 5.0 ppm Np 100 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1L. Transporting raffinate to West End Treatment Facility (WETF)	Mid-1980s–1989	3.5 ppm Pu 5.0 ppm Np 100 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1M. Denitration of SRS and/or ICPP UNH to UO ₃	1953-1970s	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1N. Maintenance on denitrators or fluid beds	1953-1970s	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1O. Conversion of SRS and/or ICPP material to UF ₄ , with reduction-hydrofluorination performed in converted lab muffle furnaces	1953-1970s	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1P. Removal of dry SRS and/or ICPP UF ₄ from process	1953-1970s	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1Q. Bomb reduction to metal	1953-1970s	0.47 ppb Pu 0.64 ppb Np 72 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1R. Sampling, fracturing, and packaging metal buttons	1953-1989	0.47 ppb Pu 0.64 ppb Np 72 ppm Tc 27.8% ²³⁶ U	Moderate
9212	1S. SRS U-Al salvage operations	1970s-1989	0.13 ppb Pu 4.9 ppb Np 1.4 ppm Tc 27.8% ²³⁶ U	Moderate

Table 2-6 Activities at the Y-12 Complex with Potential for Worker Exposure to RU

Location	Activity	Time Frame	Constituents	Occupational Exposure Potential*
9720-5	1T. Metal product shipped	1953-1990s	0.47 ppb Pu 0.64 ppb Np 72 ppm Tc 27.8% ²³⁶ U	No Significant
2. Activities Associated with Building 9206				
9206	2A. SRS UN solution “poured-up” into safe tanks	1970s-1989	0.25 ppm Pu 0.073 ppm Np 82 ppm Tc 27.8% ²³⁶ U	Moderate
9720-5	2B. SRS U-Al ingots received	1972-1990s	0.13 ppb Pu 0.49 ppb Np 1.35 ppm Tc 27.8% ²³⁶ U	No Significant
9720-5	2C. SRS dross and sweepings received	1972-1989	0.13 ppb Pu 0.49 ppb Np 1.35 ppm Tc 27.8% ²³⁶ U	No Significant
9206	2D. SRS U-Al (or dross/sweepings) dissolved in NaOH to remove Al; sodium diuranate produced	1972-1989	0.13 ppb Pu 0.49 ppb Np 1.35 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2E. SRS sodium diuranate dissolved in nitric acid to produce UN	1972-1989	0.13 ppb Pu 0.49 ppb Np 1.35 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2F. ICPP UO ₃ received, dissolved to form UN	1970s-mid-1980s	0.11 ppb Pu 4.7 ppm Np 130 ppm Tc 10% ²³⁶ U	Moderate
9206	2G. Purification of SRS and/or ICPP UN via solvent extraction (primary and secondary extraction)	1970s-1989	4.4 ppm Pu 5.9 ppm Np 190 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2H. Isolating and trucking or piping raffinate to 9212	1970s-1989	3.5 ppm Pu 5.0 ppm Np 100 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2I. Denitration of SRS and/or ICPP UN to UO ₃	1970s-1989	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2J. Maintenance on denitrators or fluid beds	1970s-1989	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2K. Conversion of SRS and/or ICPP material to UF ₄	1970s-1989	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate
9206	2L. Removal of dry SRS and/or ICPP UF ₄ from process	1970s-1989	0.5 ppb Pu 0.67 ppb Np 76 ppm Tc 27.8% ²³⁶ U	Moderate

Table 2-6 Activities at the Y-12 Complex with Potential for Worker Exposure to RU

Location	Activity	Time Frame	Constituents	Occupational Exposure Potential*
9206	2M. Bomb reduction to metal	1970s-1989	0.47 ppb Pu 0.64 ppb Np 72 ppm Tc 27.8% ²³⁶ U	Moderate
3. Activities Associated with Other Uranium Handling Facilities				
S-3 Ponds	3A. Closure of S-3 Ponds and New Hope Pond	1953-mid-1980s	0.39 ppm Pu 0.54 ppm Np 11 ppm Tc 3.0% ²³⁶ U	Moderate
WETF	3B. Treatment of nitrate waste	Mid-1980s-1990s	0.39 ppm Pu 0.54 ppm Np 11 ppm Tc 3.0% ²³⁶ U	Moderate
9720-5	3C. RU materials stored	1950s-Present	0.47 ppb Pu 0.64 ppb Np 72 ppm Tc 27.8% ²³⁶ U	No Significant

* The methodology established for the DOE Mass Balance Project considered ²³⁶U an unmonitored isotope, along with Pu, Np, and Tc. In fact, ²³⁶U is generally indistinguishable from other uranium isotopes; it has the same chemical behavior and the same dose consequences as can be seen by comparing uranium DAC values. Monitoring, both in the field and through bioassay, accounts for its presence and correctly assigns dose or risk. Other constituents, such as plutonium, are fundamentally different in that they do not have the same chemical behavior and risk. Their presence could alter the intrinsic risk of handling recycled uranium. Because ²³⁶U was monitored at the Y-12 Complex, the analysis presented in this table, which used the DOE Mass Balance Project de minimis calculation methodology, estimates the occupational exposure potential (the implied hazard) to be higher than it actually is. A calculation that considers the non-uranium, potentially unmonitored component would at times lead to the conclusion of "No Significant Occupational Exposure Potential" when ²³⁶U is more appropriately considered.

Available analytical data showed that a majority of the RU constituents of concern tended to follow the HEU through the chemical processes in Buildings 9212 and 9206. Consequently, a majority of the RU constituents ended up in the HEU metal buttons shipped to SRS. Some concentration of RU constituents (relative to the uranium mass) occurred in the various solvent extraction raffinate streams. However, calculations of potential dose using the prescribed DOE methodology indicate that the fractional contribution of the RU constituents for most process streams generally was greater than 50% (with ²³⁶U being the dominant constituent). Consequently, for most exposure scenarios identified in Table 2.6, a value of 3 was assigned for the constituent level (see tables in Appendix B).

The reader should note that the TRU-element and fission-product concentrations alone were not sufficiently high for any of the exposure scenarios to warrant this highest constituent rating of 3. Instead, the assignment of a constituent level of 3 was driven largely by the high concentrations of ²³⁶U in the SRS RU. This isotope is generally indistinguishable from the other isotopes of uranium. It has the same chemical behavior and the same dose consequence, as can be seen by comparing the uranium derived air concentrations (DAC) limits. For example, the DAC for Class W ²³³U, ²³⁴U, ²³⁵U, ²³⁶U,

and ^{238}U is the same—i.e., $3\text{E-}10$ microcuries per milliliter ($\mu\text{Ci/ml}$). Similarly, the dose conversion factors are also the same. Monitoring, both in the workplace and through bioassay, accounted for the presence of ^{236}U and correctly assigned dose or risk. This approach was based on two factors:

- Air sampling in the workplace was retrospective via filter collection with subsequent gross alpha counting on the filter. As such, all alphas were counted and would have included those from ^{236}U . Because the DAC is the same for all uranium isotopes of concern, the need for personnel protection would have been evaluated with all radioactivity appropriately considered. The only exception would have been that the alphas counted associated with any transuranic present would have been attributed to uranium. This was considered during the development of the acceptance criteria for RU (see Section 4.3).
- In terms of bioassay monitoring, the analytical method (fluorometric procedure) measured total uranium. As a result, ^{236}U was considered in the overall dose assessment. To be conservative, the uranium result was all attributed to ^{234}U , which has the highest specific activity of the uranium isotopes of concern. However, using the methodology prescribed by the DOE Project Plan, ^{236}U is included in the calculation as an additional RU constituent. Because ^{236}U was monitored and accounted for, its inclusion as a constituent distorts the implied hazard. A calculation that more appropriately treats ^{236}U in considering the non-uranium, potentially unmonitored component would at times instead lead to the conclusion of “No Significant” OEP.

In contrast to the SRS RU with high ^{236}U content, ICPP RU had an average ^{236}U content of 10%. Activities involving only ICPP RU thus received a constituent level rating of 2.

Airborne potential values associated with the various exposure scenarios ranged from 0 to 3. The lowest airborne rating was assigned to HEU operations in which there was virtually no potential for direct worker contact with RU. A value of 1 was assigned to HEU operations involving direct exposure to metal or consolidated solids. A value of 2 was assigned for activities involving exposure to liquid solutions that might spray or evaporate to dryness outside the equipment. A value of 3 was assigned to operations involving direct contact with finely divided RU solids. Duration exposure values were based on actual contact time with RU as defined by DOE (see Appendix B).

Most of the potential exposure activities at the Y-12 Complex were found to have a “Moderate” OEP rating as a result of the combined product of a constituent level value of 3 for Savannah River RU or a value of 2 for Idaho RU with a value of 1 or 2 for airborne potential and exposure duration. Certain maintenance activities involving equipment that contained finely divided RU solids were assigned a value of 3 for airborne potential. However, because these types of maintenance activities were not performed very often, the overall OEP was rated “Moderate,” with a cumulative score of 9.

In no instance did any identified activity involve a combination of airborne potential, constituent level, and exposure duration factors that produced an OEP score in the “High” range. Although some activities presented moderate OEP scores, the average

DAC for the areas associated with RU was on the order of only 3% of the Plant Action Level (PAL).

The following provides information on the activities listed in Table 2.6. The numbering system used in the table (i.e., 1A, 1B, etc.) is also used below.

1A. ICPP UN Solution Received in Safe Bottles: UN solution was received from ICPP in safe bottles from 1953 to the early 1970s. These solutions were weighed, sampled for U-content, uranium isotope distribution, and RU components. The uranium was removed from the UN solution by peroxide precipitation. The receiving and processing steps to establish accountability were performed in well-ventilated hoods resulting in “no significant” OEP.

1B. SRS Tanker Truck Weighed for Gross Weight: UN was received in tanker trucks (3,800 – 5,000 gallon capacity) with ^{235}U concentration of 5 g/liter. The tankers were gross weighed at Building 9929-1, and the solution was transferred by pump from the tanker into a storage tank in Building 9812 in the 9212 Complex. A tare weight was obtained for the empty tanker at Building 9929-1 prior to return to SRS. This operation had “no significant” OEP.

1C. SRS Material Sampled: In Building 9812, the UN solution circulated for 3 hours and was then sampled for U-content, uranium isotope distribution, and RU components. This operation was performed with pumps and enclosed piping, resulting in “no significant” OEP.

1D. ICPP UN Solution Poured into “Pour-Up” Stations: The UN solution received in safe bottles from the ICPP was transferred at the “pour-up” station from the bottles to the storage tanks. The transfer was performed in well-ventilated hoods and was considered to have “low” OEP.

1E. SRS UN Solution Pumped to 9212: The sampled UN solution in Building 9812 was transferred by pump to the evaporator feed tanks in Building 9212. This transfer of UN solution through closed piping with an operator in attendance resulted in “no significant” OEP.

1F. SRS and/or ICPP UN Evaporated and Concentrated: UN solution received from SRS was evaporated to concentrate the uranium to approximately 150 – 200 g/liter. UN received from ICPP was already concentrated. This concentrated UN was relatively pure and was pumped directly to secondary extraction. The OEP for this process was considered “moderate” due to the high uranium content and worker time exposure.

1G. Manual Filling and Loading of SRS and/or ICPP UN into Safe Bottles for Transfer to 9206: The concentrated UN solution was manually drained from storage tanks into tare-weighed safe bottles, capped, gross weighed, and placed in a 6-bottle dolly for transfer to Building 9206. Prior to draining the concentrated UN into safe bottles, the UN was thoroughly mixed in the storage tanks, and samples were removed to determine uranium accountability for the transfer between Buildings 9212 and 9206. The OEP was

considered “moderate” due to the high uranium content in the UN and the manual handling of the safe bottles.

1H. ICPP UO_3 Received, Dissolved to Form UN: UO_3 received from ICPP was weighed and sampled to determine U-content, uranium isotope distribution, and RU components. The UO_3 was then dissolved in HNO_3 to prepare concentrated UN solution ready for secondary extraction. The UO_3 was measured for accountability in a glove box. The dissolution was performed in a well-ventilated hood. These processes were considered to have “moderate” OEP.

II. Purification of SRS and/or ICPP UN via Solvent Extraction: Purification of UN from SRS and ICPP consisted of two extraction processes: primary and secondary.

The relatively pure, concentrated UN solutions from the evaporator feed tanks were first processed through secondary extraction. The organic solvent in this case was tributyl phosphate (TBP). The organic was passed counter current through the UN solution in a vertical pulsed plate column. The uranium was absorbed by the organic solvent and then removed from the solvent with demineralized water. The uranium solution was collected in storage tanks for further processing. The secondary extraction raffinate containing 2-5 wt % uranium was recycled and became part of the feed stream for primary extraction.

The dilute uranium solutions, after filtration, evaporation, and addition of aluminum nitrate [$\text{Al}(\text{NO}_3)_3$], were processed through primary extraction. The organic solvent, dibutyl carbitol, was passed counter current through the dilute uranium aqueous solution in a series of vertical columns with pulse plates. The uranium was absorbed into the organic phase. The uranium was then removed from the organic phase with dilute HNO_3 and water. This was accomplished by passing the organic phase counter current to the aqueous stream in a second series of vertical columns with pulse plates. This dilute UN solution was transferred to evaporator feed tanks where it was concentrated. The primary extraction raffinate, containing approximately 1 ppm uranium, was collected in tanker trucks and taken to Building 9818 for waste treatment.

These processes were considered to have a “moderate” OEP.

1J. Discard of Solvent Extraction Raffinate to S-3 Ponds: The primary extraction raffinate containing approximately 1 ppm uranium was processed in Building 9818 to recover $\text{Al}(\text{NO}_3)_3$ for reuse. This was achieved by evaporation to a heavy sludge and the solids removed by centrifuge. The raffinate was then processed through a bioreactor before transfer to the S-3 Ponds (until their closure in 1984). Nitric acid (HNO_3) removed during the evaporation was combined with HNO_3 recovered from the evaporator and other condensates generated in the chemical processes. This activity was considered “moderate” for OEP since, while the uranium content was low, some RU constituents remained.

1K. Feeding of Raffinate to 9212 Bioreactor: After the $\text{Al}(\text{NO}_3)_3$ was removed from the primary extraction raffinate, the raffinate was transferred into the bioreactor tank. An equal volume of calcium acetate/nutrient was added for the biological decomposition of

the remaining HNO_3 . This activity was considered “moderate” for OEP since, while the uranium content was low, some RU constituents remained.

1L. Transporting Raffinate to West End Treatment Facility (WETF): After closure of the S-3 Ponds in 1984, the raffinate from the primary extraction process was transferred by tank truck to the WETF. This activity was considered “moderate” for OEP since, while the uranium content was low, some RU constituents remained.

1M. Denitration of SRS and/or ICPP UNH to UO_3 : The secondary extraction product was concentrated in an evaporator to molten uranyl nitrate hexahydrate (UNH) and stored in a steam-jacketed tank to prevent solidification. Molten uranyl nitrate was conditioned by addition of 1,500 ppm sulfuric acid. This resulted in a more chemically reactive product upon conversion to UO_3 . Conditioned molten uranyl nitrate was denitrated by pumping the uranyl nitrate into a five-inch diameter, heated stirred-trough reactor, which produced UO_3 . The UO_3 was in the form of freely flowing spherical particles with a predominant size range of –30 mesh to +100 mesh (U.S. sieve size). As molten uranyl nitrate was continuously pumped into the heated stirred-trough reactor, the UO_3 product overflowed into a receiver tank. The OEP for this process was considered “moderate.”

1N. Maintenance on Denitrators or Fluid Beds: With the exception of emergencies, maintenance was usually performed during the scheduled inventory shutdown period. Any maintenance requiring opening the denitrators or fluid beds was carefully planned to avoid potential health physics problems associated with uranium airborne exposure. This activity was considered to have “moderate” OEP.

1O. Conversion of SRS and/or ICPP Material to UF_4 : Uranium trioxide was converted to UF_4 in a two-step fluid-bed process. First, UO_3 was hydrogen-reduced to UO_2 in a stainless steel fluidized-bed reactor. The UO_2 was transferred to an Inconel fluidized-bed reactor and converted to UF_4 with anhydrous hydrogen fluoride. Heat was supplied to both reactors by external clam-shell electrical resistance heaters. Both reactor off-gas systems contained micrometallic filters backed up in series by porous carbon filters and were equipped with gamma monitors to detect filter failure. These processes were performed in closed systems, and powder transfers were achieved via vacuum and pneumatic gas flows. The OEP was considered “moderate.”

1P. Removal of Dry SRS and/or ICPP UF_4 from Process: In the early years (1953 – late 1960s) before the installation of denitrators and fluid beds, the impure UN was combined with hydrogen peroxide and the resulting uranium peroxide was converted to UF_4 using platinum trays and muffle furnaces. In another batch process, purified ammonium diuranate was precipitated from UN with the addition of ammonium hydroxide and converted to UF_4 as described above. All of these processes were performed manually. This activity was considered to have “moderate” OEP due to its reliance on manual handling and processing. After the denitrators and fluid beds were installed, manual handling of the compounds was replaced with pneumatic transfer.

1Q. “Bomb” Reduction to Metal: The UF_4 was converted to uranium metal, referred to as “buttons” (Fig. 2.6-1), which derive their shape from the bottom of the crucible in which they solidify. The UF_4 was converted to metal by “bomb” reduction through reaction with calcium at high temperature. Granular calcium metal was mixed with the UF_4 and loaded into a stainless steel reactor (induction-heated furnace) fitted with a calcium fluoride liner, or crucible. CaF_2 sand was used as backfill between the crucible and the reactor wall.

Along with a mixture of UF_4 and calcium, the reactor was also charged with a lithium “biscuit” and an igniter capsule. While the reactive metals, lithium and calcium, will both reduce UF_4 , calcium served as the primary reductant for the process. Lithium was added to lower the melting point of the slag product from the reaction by taking advantage of the calcium fluoride – lithium fluoride eutectic. The CaF_2 -LiF slag produced had a lower melting point than either CaF_2 or LiF. This lower melting-point slag allowed for cleaner separation of the metal button from the slag. The igniter capsule aided initiation of the reduction reaction by providing a small exothermic reaction and associated heat spike.

This processing was performed in glove boxes and well-ventilated hoods. The OEP was considered to be “moderate.”



Fig. 2.6-1 Metal button.

1R. Sampling, Fracturing, and Packaging Metal Buttons: Uranium metal buttons produced were cleaned with acetic acid, dried, weighed, and transferred to Building 9212; four buttons/batch were identified with the percentage of ^{235}U of the UF_4 greensalt blend. A composite sample of the four buttons was submitted to the laboratory for U content, uranium isotope distribution, and 32-element specifications. If the percent ^{235}U was within 0.3% of the UF_4 blend, the data was acceptable. Every tenth batch of four buttons was analyzed for RU components. Each of the four buttons was then fractured or sheared into small pieces as specified by SRS. The uranium metal pieces were packaged into DOE-approved containers and transferred to storage in Building 9720-5 to await shipment to SRS. These operations were performed inside well-ventilated hoods or glove boxes; the OEP was considered “moderate.”

1S. SRS U-Al Salvage Operations: Uranium/aluminum alloy received from SRS was processed first by dissolution of the aluminum with NaOH. The sodium diuranate solids recovered by filtration were then dissolved in HNO_3 . This dilute UN solution and the insoluble solids were sampled to establish uranium accountability. The spent NaOH filtrate was transferred to waste treatment. This process was transferred to Building 9206 in September 1983. This activity was performed in well-ventilated hoods. The OEP was considered “moderate.”

1T. Metal Product Shipped: Uranium metal pieces were stored in Building 9720-5 until SRS requested shipment. The metal was stored in closed containers and presented “no significant” OEP.

2A. SRS UN Solution “Poured-Up” into Safe Bottles: SRS UN solution received in safe bottles from 9212 was check weighed, and the UN was transferred to the secondary extraction feed tanks. This transfer was performed via manual pour-up or by vacuum. Although the uranium concentration of the solution was high, this transfer activity was considered to have “moderate” OEP.

2B&C. SRS U/Al Ingots, Dross, and Sweepings Received: Beginning in late 1983, U-Al alloy ingots, dross, and floor sweepings were received in Building 9206 for uranium recovery. Prior to this time, Building 9212 received this material. Receipt of U-Al ingots was considered to have “no significant” OEP.

2D&E. SRS U/Al Dissolved in NaOH to Remove Al; Sodium Diuranate Dissolved: Beginning in September 1983, this process was transferred to Building 9206. Uranium/aluminum alloy received from SRS was processed first by dissolution of the aluminum with NaOH. The sodium diuranate solids recovered by filtration were then dissolved in HNO₃. This dilute UN solution and the insoluble solids were sampled to establish uranium accountability. These activities were performed in well-ventilated hoods. From the mid-1980s, the spent NaOH filtrate was transferred to waste treatment. The OEP was considered “moderate.”

2F. ICPP UO₃ Received, Dissolved to Form UN: After 1971, UO₃ from ICPP was received in Building 9212, and accountability was established for U-content, uranium isotope distribution, and RU components. In the late 1970s to the late 1980s, the UO₃ was processed in Building 9206. The dissolution was performed in a well-ventilated hood. These processes were considered to have “moderate” OEP.

2G. Purification of SRS and/or ICPP UN: These processes are described in activity 1I.

2H. Isolating and Trucking or Piping Raffinate to 9212: Primary extraction raffinate was collected in a tanker and trucked to Building 9818. This raffinate did not have the Al(NO₃)₃ removed. It was pumped into the bioreactor along with the 9212 primary extraction raffinate, after which the Al(NO₃)₃ was removed. This raffinate, while low in uranium content, contained RU constituents and was considered to have “moderate” OEP.

2I. Denitration of SRS and/or ICPP UNH to UO₃: The secondary extraction product was concentrated in an evaporator to molten uranyl nitrate hexahydrate (UNH) and stored in a steam-jacketed tank to prevent solidification. Molten UNH was conditioned by addition of 1,500 ppm sulfuric acid. This resulted in a more chemically reactive product upon conversion to UO₃. These processes were considered to have “moderate” OEP.

2J. Maintenance of Denitrators and Fluid Beds: With the exception of emergencies, maintenance was usually performed during the scheduled inventory shutdown period. Any maintenance requiring opening the denitrators or fluid beds was carefully planned to avoid potential health physics problems associated with uranium airborne exposure. This activity was considered to have “moderate” OEP.

2K. Conversion of SRS and/or ICPP Material to UF₄: Uranium trioxide was converted to UF₄ in a two-step fluid-bed process. First, UO₃ was hydrogen-reduced to UO₂ in a stainless steel reactor. The UO₂ was pneumatically transferred to an Inconel reactor and hydrofluorinated to UF₄ with gaseous anhydrous hydrogen fluoride. Heat was supplied to both reactors by external clam-shell electrical resistance heaters. Both reactor off-gas systems contained micrometallic filters backed up in series by porous carbon filters and were equipped with gamma monitors. The OEP was considered “moderate.”

2L. Removal of Dry SRS and/or ICPP UF₄: The UF₄ produced by the two-stage fluid beds was removed from the process by pneumatic transfer to a vertical safe receiver. The UF₄ was sampled for U-content and uranium isotope distribution, and stored awaiting reduction to metal. The pneumatic transfer from the closed equipment into the glove boxes presents only “moderate” OEP.

2M. “Bomb” Reduction to Metal: The UF₄ was converted to uranium metal, referred to as metal “buttons,” which take their shape as they solidify from the shape of the bottom of the crucible in which they are formed. The UF₄ was converted to metal by “bomb” reduction with calcium. Granular calcium metal was mixed with the UF₄ and loaded into a stainless steel reactor (induction-heated furnace) fitted with a calcium fluoride liner, or crucible. CaF₂ sand was used as backfill between the crucible and the reactor wall.

Along with a mixture of UF₄ and calcium, the reactor was also charged with a lithium “biscuit” and an igniter capsule. While the reactive metals, lithium and calcium, both reduce UF₄, calcium served as the primary reducer for the process. Lithium was added to lower the melting point of the slag product by taking advantage of the calcium fluoride – lithium fluoride eutectic. The CaF₂-LiF slag produced had a lower melting point than either CaF₂ or LiF. This lower melting-point slag allowed for cleaner separation between the metal product and the slag, and thereby produced a sound, smooth metal button that separated easily. The igniter capsule aided initiation of the reduction reaction by providing a small exothermic reaction and associated heat spike.

This processing was performed in closed equipment (glove boxes) and well-ventilated hoods. The OEP was considered to be “moderate.”

3A. Closure of S-3 Ponds and New Hope Pond: Closure of the S-3 Ponds was accomplished by neutralizing the wastewater to precipitate the RU components and to allow denitrification prior to pumping the liquid off through an NPDES discharge point and leaving the contaminated sludge exposed. A gravel, clay, and rubber membrane and asphalt cap was placed over the ponds to complete the closure. These closure activities presented only a “moderate” OEP.

Closure of New Hope Pond was performed in a similar manner to the S-3 Pond closure, with the exception that the New Hope Pond sludge was removed before the cap

was installed. Also, the cap construction did not include an asphalt layer. Since the pond sludge contained significantly smaller amounts of RU constituents than the S-3 Pond sludge, the OEP for the closure was less than that assigned to the S-3 Pond closure.

3B. Treatment of Nitrate Waste: Nitrate wastewaters from the UN solution evaporators and raffinates from the solvent extraction systems were periodically transported to the WETF for removal of the nitrate and final treatment prior to discharge under NPDES permit to East Fork Poplar Creek. The acid streams were first pumped into several large stirred tank reactors for batch biodenitrification. The HEU wastewaters were mixed with various aqueous waste streams containing depleted uranium generated elsewhere in the Y-12 Complex and neutralized with caustic. Carbon nutrients were subsequently added to the tanks to initiate and sustain the biological process. After biodenitrification, the resulting liquid and suspended solids were pumped to the WETF for pH adjustment, flocculation, and filtration. Essentially all of the process uranium (both enriched and depleted) and RU constituents were precipitated and collected with the process solids. The resulting semi-dried solids were pumped as a thick slurry to a dedicated set of large-volume tanks for long-term storage. Operators that worked around the solids collection, drying, and transport steps of the process were most likely to be exposed to the RU constituents. Because the HEU-derived streams were substantially diluted with depleted uranium from other plant operations, RU concentrations (expressed on a total uranium basis) were low. Further, the contaminated solids were not dried beyond a pumpable solid slurry and were not easily dispersed. Hence, the WETF operation was rated as having only a “moderate” OEP.

3C. RU Materials Stored: The material is stored in closed containers and so has no airborne potential, thus presenting “no significant” OEP.

2.7 WORKER RADIOLOGICAL PROTECTION PROGRAMS

Extensive documentation of various radiological protection programs beginning in the early 1950s was identified and reviewed by the Project Team. The documentation provides evidence of health physics programs that included personnel monitoring, urinalysis, process area monitoring and contamination control, plant site and off-site monitoring and contamination control, and special surveys.³ Biannual Health Physics Progress Reports document the issuing of film badges, finger rings, special badges, and special rings or pads, and neutron film badges.⁴ Beginning around 1960 and through the 1970s, the Health Physics and Industrial Hygiene Sections were organizations under the Radiation Safety Department, which was responsible for issuing the *Y-12 Radiation Safety Manual*.⁵ The following sections summarize the contents of these and other documents reviewed for this project.

³ Union Carbide Nuclear Company, *The Y-12 Health Physics Program*, 1957.

⁴ Carbide and Carbon Chemicals Company, *Health Physics Progress Reports*, 1953.

⁵ Union Carbide Nuclear Company, *Y-12 Radiation Safety Manual*, 1963.

Roles and Responsibilities

As stated in the 1963 *Y-12 Radiation Safety Manual*, responsibility for the protection of the employee against radiation health hazards rested with the line organization to the same extent that line-organization personnel were responsible for plant operation, production, and research. While the primary responsibility for implementing safety policy rested with line supervision, staff and service groups were established to provide technical assistance, to render service in the investigation and evaluation of radiation and industrial-hygiene problems, to maintain exposure records, and to give proper radiation-worker training to employees.

Responsibilities of the line organization included:

- informing the Radiation Safety Department of potentially hazardous processes or materials being contemplated or used and initiating requests for protective devices or services;
- formulating, administering, and enforcing safety rules and regulations necessary to the health physics and industrial hygiene programs in all areas within the scope of their authority;
- planning, incorporating, and utilizing adequate health safeguards and practices in new equipment and/or procedures;
- informing all concerned employees of potential health hazards and the necessary safeguards established to guard against them;
- arranging for participation of employees in established personnel monitoring programs;
- maintaining material control by the proper routing, shipping, and disposal of contaminated materials in accordance with established procedures;
- determining whether company clothing would be made available and whether it was mandatory that clothing be worn for contamination or exposure control; and
- issuing Safety Work Permits to maintenance supervision.

The employee was expected to follow rules and regulations pertaining to job hazards for his location and assignment, monitor his person and work area as required, and notify the immediate supervisor of any known exposure to radioactive materials or conditions exceeding the allowable radiation or contamination values.

Staff Groups consisted of the Laboratory Division, Safety Department, Medical Department, and the Radiation Safety Department (which included health physics and industrial hygiene). Radiation Safety Department responsibilities included the following functional activities:

- providing line supervision with technical assistance in the establishment of suitable environmental controls, carrying out an effective environmental monitoring program for substances of concern, and recommending appropriate equipment, systems, and analytical procedures;
- continually evaluating potential personnel exposures by means of external monitoring, body fluid or excreta analyses, in vivo counting, and X-ray and clinical examinations (Medical Department), and maintaining suitable records and issuing

reports to apprise management of existing conditions and/or immediate action requirements;

- providing technical information, assistance, and guidance to ensure conformance to then-AEC regulations and other federal and state laws pertaining to these functions, namely, employee exposure records, transportation of hazardous materials, waste disposal, release of effluents to the public domain, and exposure of the general population;
- auditing operations for compliance with prescribed procedures, such as (1) advising appropriate supervision of violations and, if necessary, taking immediate action through line supervision to have the operation shut down and (2) seeking improved methods of reliability as well as recommending equally safe methods of improved operating efficiency;
- conducting plant-wide meetings, preparing and issuing useful reference and training materials, assisting in emergency preparedness planning, and offering consultation on immediate problems;
- reviewing proposed alterations, modifications, or additions to plant facilities and equipment for compliance with pertinent plant health and safety standards;
- assisting investigation of conditions in work areas that may be suspected of contributing to the health problems of employees, upon the request of the Medical Department;
- providing special services to other departments within the plant, such as (1) sampling and analyzing potable water and sewer effluents to evaluate the control of waste discharge and to determine the possibility of potable water contamination, (2) sampling stack gases for operations supervision to determine what material is safe to be released to the atmosphere, and (3) recommending shielding requirements for the safe use of radioactive sources and X-ray units.

Plant Operational Guides

Protection guides used in administering the radiation safety and industrial hygiene programs followed those established by the Federal Radiation Council, the National Committee on Radiation Protection, the International Commission on Radiological Protection, the American Industrial Hygiene Association, and others.⁶ Plant limits and guidelines included:

- Radiation Protection Guides (RPG) for exposure to external radiation (penetrating, skin, and extremities) and
- RPG for internal exposure (maximum permissible body burdens and concentration in urine for uranium, neptunium, plutonium, thorium, tritium, and other isotopes).

Personnel monitoring at the Y-12 Complex was accomplished primarily through the use of film badges and/or rings for external exposures and bioassay and in vivo counting for internal exposures. Control and action points, including additional sampling and work

⁶ Union Carbide Nuclear Company, *Y-12 Radiation Safety Manual*, 1963.

restrictions, were included in the RPGs and were described as follows in a 1962 report documenting a review of the Y-12 Complex health protection programs.⁷

“Actions taken at the following levels of exposure include: (1) quarterly reports to supervisors indicating the number of their people who exceed 300 mrem/quarter penetrating radiation and 1,000 mrem/quarter non-penetrating radiation, (2) quarterly reports to supervisors naming the people who exceed 1.25 rem penetrating and those who exceed 7.5 rem non-penetrating for the quarter, (3) removal from radiation areas is recommended for those who exceed 3 rem/quarter penetrating or 10 rem/quarter non-penetrating radiation. Such a restriction would be lifted only when a consecutive 4-quarter exposure drops below 5 rem penetrating or 30 rem non-penetrating; (4) removal from radiation areas is recommended for those whose average annual exposure exceeds 5 rem penetrating radiation, and they would be allowed to return only when the cumulative exposure during Y-12 Complex employment averages less than 5 rem/year for penetrating radiation, regardless of the individual’s previous radiation history.

“Monitoring for internal exposure to uranium routinely involves 1,800 employees. About 10% of these are sampled weekly, 30% monthly and 60% quarterly. The criteria for action taken at various urine concentrations are detailed and well documented. Actions taken at the level of significant internal exposure are usually based on concurrent in vivo measurements, however, definite indication of a body burden by either method is sufficient to initiate investigative or restrictive action depending on the level involved. The frequency of sampling is determined semiannually based on a statistical evaluation of results from the previous six months. All urine analyses are made by the Laboratory Development Department and the results sent weekly to RSD (Radiation Safety Department). In case of an unusually high sample, RSD is notified immediately.”

Additional Radioactivity Concentration Guides (RCG), Plant Action Limits (PAL), and controls were established:⁸

- concentration guides for materials in air (including uranium, neptunium, and plutonium),
- concentration guides for toxic materials in water (including uranium, neptunium, and plutonium),
- control criteria for surface contamination (including uranium, neptunium, and plutonium), and
- control criteria for shipments leaving the Y-12 Complex (including uranium and plutonium).

Workplace air analyses were performed and divided into three categories: operational and breathing zone, general air, and outside air monitoring.⁹

Operational and breathing zone samples were taken to determine the airborne contamination generated by specific operations and/or to estimate the amount that an employee might breathe during a specific time. Health physics recommendations were made on the basis of these samples for effective personnel precautions and various

⁷ *Review of Y-12 Plant Health Protection Programs*, correspondence from S. R. Sapirie, ORO Manager to Dr. C. E. Larson, Vice President, Union Carbide Nuclear Company, September 26, 1962.

⁸ Union Carbide Nuclear Company, *Y-12 Radiation Safety Manual*, 1963.

⁹ Union Carbide Nuclear Company, *Y-12 Plant Quarterly Health Physics Report*, September 8, 1964.

administrative and mechanical controls. During the second quarter of 1964, uranium samples numbered 1,191 with an additional 1,894 samples obtained by permanently installed operational samplers for uranium analysis and 1,637 for thorium determination.

The overall exposure potential of any particular operation is not only a function of the concentration but also of the frequency and time required for the operation. The quarterly report suggests that priority be given to the jobs which have the highest product of (concentration) x (time of operation) x (frequency of operation). Weekly Air Concentration Indices (WACI) for specific operations were calculated as follows:

$$WACI = \text{Concentration (dpm/m}^3\text{)} \times \text{length of each operation (min)} \times \text{(number of times operation performed per week)} \times 0.00042 \text{ (conversion factor)}$$

The conversion factor was used to make the magnitude of the number comparable with the PAL of 70 dpm/m³. The WACI calculation means that performing the operation without respiratory protection is equivalent in exposure potential to breathing air for the entire work week at the concentration indicated. High-uranium air concentrations make it necessary to require the use of respiratory protective equipment in the immediate area of the operations being performed. At those times, it was recommended that respiratory protection be worn on all operations exceeding 200 dpm/m³. It was noted in the report that “such practice is being followed at most such locations.”

General air sampling was performed to determine average airborne contamination from both uranium and thorium in several work areas of the plant. These included Metal Preparation (Buildings 9212, 9215, and 9206); Development (Buildings 9212 and 9202); Maintenance (Building 9206); Fabrication (Building 9206); and Technical Services (various buildings). At that time, the PAL for uranium was 70 dpm/m³ and the PAL for thorium was 4.4 dpm/m³. All areas during the reporting period were below the PAL, although four individual samplers in some areas averaged above the PAL for uranium.

Eleven outside air monitors, located in relation to various process buildings and prevailing winds in the Y-12 Complex area, were operated continuously. The filters were changed and analyzed biweekly for gross alpha and beta activity. All readings during the reporting period were below the PAL.

2.8 ENVIRONMENTAL IMPACT OF RECYCLED URANIUM CONSTITUENTS

Various sources that documented the potential environmental impact of RU components from the Y-12 Complex and the Oak Ridge Reservation were identified and reviewed by the Project Team. These reports are summarized in the following sections.

2.8.1 Historical Radionuclide Releases from Current DOE-ORO Facilities

An ORO report titled *Historical Radionuclide Releases from Current DOE Oak Ridge Operations Office Facilities*, OR-890, May 1988, documents uranium and some radionuclide releases to the air and water and burial of solid waste. This report is summarized below.

History of Airborne Emissions from the Y-12 Complex

The major source of airborne radiological emissions from the Y-12 Complex has historically been, and continues to be, emissions of small uranium particles from metal-machining and chemical-processing operations. The primary means of controlling these emissions is the use of High Efficiency Particulate Air (HEPA) filters, baghouses, and exhaust gas scrubbers. The 13.7 curies of uranium emissions from the Y-12 Complex from 1944 to 1986 resulted primarily from major enriched uranium sources. Uranium emission information after 1954 was obtained from Y-12 Complex accountability records, the DOE Effluent Information System Radioactivity Summary Report, and the Solid Waste Information Management System. Prior to 1954, analytical and sampling techniques at the Y-12 Complex were not able to detect airborne sources of uranium, but enough data was identified in health physics reports and other sources to make some emissions estimates in the report. Since data was not available from the time period of 1948 to 1953, emissions estimates for that time period were not made.

Uranium emissions from the Y-12 Complex were highest from 1959 through 1970. This can generally be attributed to increases in production during that time. The construction of new baghouses and other equipment at the Y-12 Complex beginning in 1969 improved the control of uranium particles and lowered overall plant emissions. From 1984 to 1986, several major enriched uranium emission control systems at the Y-12 Complex were upgraded to further reduce emissions (as part of the Production Capabilities Restoration Project). Additional reductions in emissions were achieved as the Air and Water Pollution Control Project was completed in 1988 with the installation of additional emission controls.

History of Liquid Effluents from the Y-12 Complex

Liquid effluent releases of radioactivity from the Y-12 Complex have generally been uranium solutions from the same sources that produced airborne emissions. In addition, sources of contamination, such as outside storage facilities, allowed runoff of precipitation containing uranium. Liquid wastes containing economically recoverable HEU have historically been recycled in Y-12 Complex production operations. Liquid wastes that did not contain recoverable HEU were discarded. Until the early 1980s, wastewater treatment facilities were not generally available, and so the waste was discharged into the storm sewer system and from there into East Fork Poplar Creek. Beginning in 1951 and until about 1983, some liquid wastes containing both enriched and depleted uranium were discharged into the S-3 Ponds located in the western end of the Y-12 Complex site. Leakage from the S-3 Pond area contributed to uranium releases into Bear Creek, as did precipitation runoff from the Bear Creek Burial Grounds (BCBG), which were used to dispose of depleted uranium solid waste.

In March 1984, when ORGDP received a permit to process Y-12 Complex aqueous waste, the discharge of process wastewater into the S-3 Ponds was discontinued. The wastewater contained in the ponds at the time of closure was treated to remove contaminants and was discharged under the Y-12 Complex NPDES permit.

History of Contaminated Solid Waste Disposal at the Y-12 Complex

Radioactive solid wastes generated from the various Y-12 Complex production processes include uranium and uranium-contaminated materials. Uranium wastes include depleted uranium metal and oxide in the form of chips, turnings, powders, scrap, and process residues along with uranium contamination resulting from the milling and machining processes. These process residues consist of uranium-contaminated materials, such as gloves, floor sweepings, filters, and demolition debris.

Most of the solid wastes have been buried in the BCBG, while some were deposited in burial areas within the plant perimeter fence and on Chestnut Ridge. Because most of the buried uranium waste is depleted uranium metal chips, and since this metal can ignite spontaneously, the chips were placed in dumpsters that contained water to prevent spontaneous burning. The dumpsters containing both uranium and water were weighed prior to burial. Because the weight of uranium shown in disposal records is actually the total weight of the depleted uranium and the water together, the solid waste report numbers are high due to the water weight. This positive bias resulted in an error in the quantities reported in the 1985 uranium release report of approximately 1,500,000 kg of depleted uranium from 1947 to 1984.

Summary of Radionuclides Released from the Y-12 Complex

Uranium releases from the Y-12 Complex between 1944 and 1987 were summarized in OR-890 as follows:

- Air 6,296 kg
- Water 182,374 kg
- Burial 17,290,523 kg

Although the most significant releases have been uranium, the DOE report documents some release of technetium. Prior to 1972, liquid wastes containing uranium that were transferred to the S-3 Ponds were recorded as burials. Approximately 2,680 grams of technetium were received from ORGDP and directly disposed of in the ponds as contaminated aqueous waste. Other radionuclides in the waste stream associated with the processing of reactor product uranium solutions also likely went to the S-3 Ponds (although recorded as burials). Since measurements were made for contamination control purposes only, the exact quantities of material that went to the ponds are unknown. Reporting thresholds were established for these materials for accountability and security purposes. Releases to the ponds were always below these reporting thresholds.

2.8.2 Environmental Radioactivity Levels News Releases

Quarterly news releases on Environmental Radioactivity Levels at the Oak Ridge Gaseous Diffusion Plant from 1959 through 1964 report data gathered from air monitoring (atmospheric contamination by long-lived fission products and alpha-emitting

materials), water monitoring, and gamma measurements.¹⁰ While these news releases were published by ORGDP, the data were gathered for the entire Oak Ridge Reservation, thus including releases from ORGDP, ORNL, and the Y-12 Complex, as well as off-site sources (e.g., Kingston Steam Plant prior to enactment of clean air legislation in the early 1970s).

Air Monitoring

Atmospheric contamination by long-lived fission products and fallout occurring in the general environment of East Tennessee were monitored by two systems of stations during this period. One system consisted of seven stations that encircled all the plant areas and provided data for evaluating the impact of all Oak Ridge Operations on the immediate environment. A second system consisted of eight stations encircling the Oak Ridge area at distances of 12 to 120 miles; after 1961, only seven stations were active.¹¹

Sampling was accomplished by passing air continuously through filter paper. The data collected were accumulated and tabulated in average $\mu\text{Ci/cc}$ of air sampled. Figures 2.8-1 and 2.8-2 show the locations of both the perimeter and remote continuous air monitoring stations. Summaries of the data for the perimeter and remote stations are shown in Tables 2.8-1 and 2.8-2.

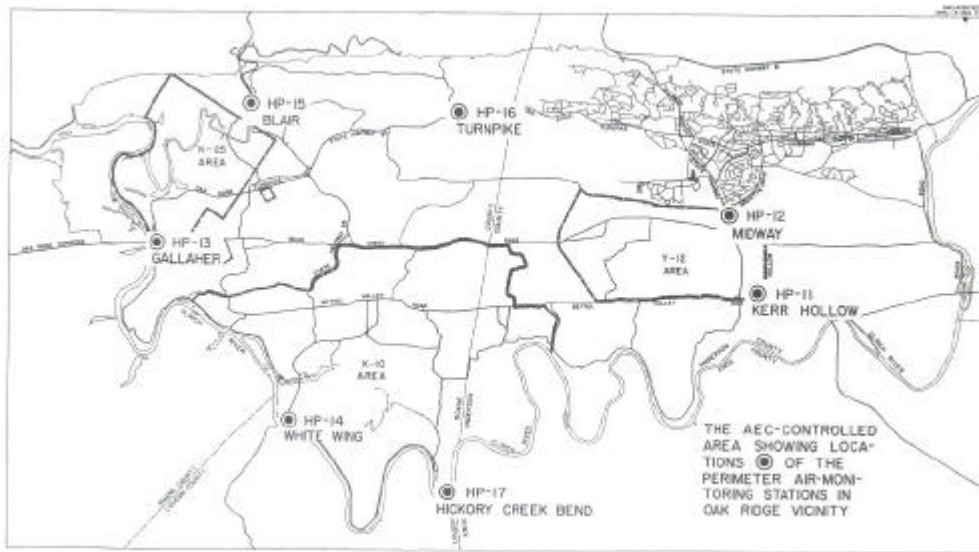


Fig. 2.8-1 Continuous Air Monitoring Data – Perimeter Stations.

¹⁰ News Releases, *Environmental Radioactivity Levels, the Oak Ridge Gaseous Diffusion Plant, ORGDP*, January 1959 through June 1964.

¹¹ The Berea, Kentucky, remote station provided no samples after 1961.

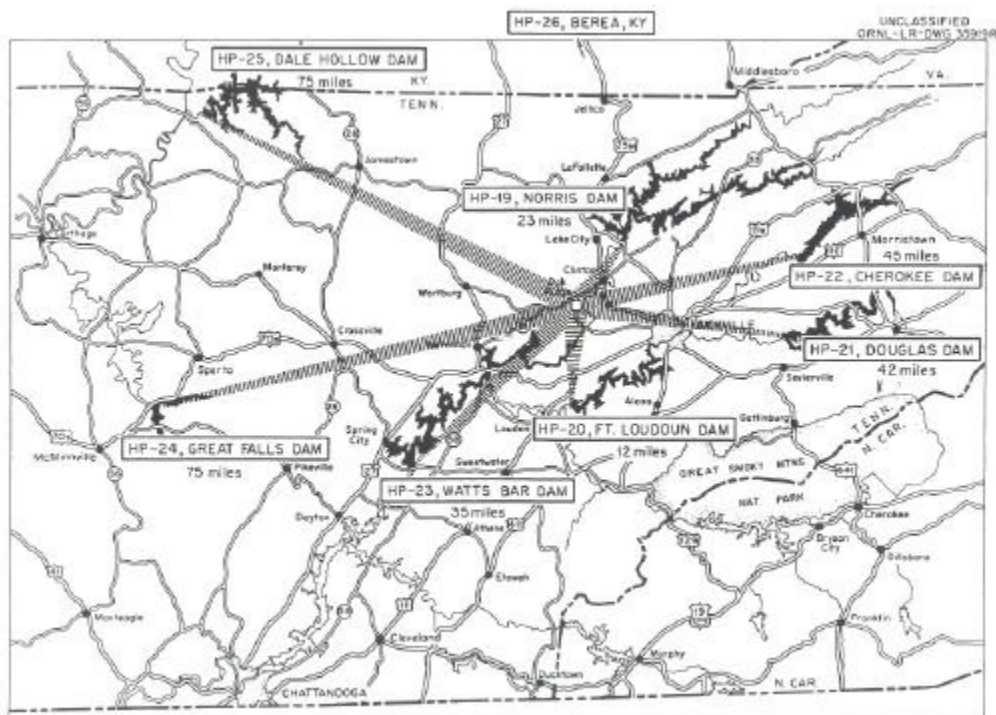


Fig. 2.8-2 Continuous Air Monitoring Data – Remote Stations.

Table 2.8-1 Continuous Air Monitoring Data – Perimeter Stations
Long-Lived Gross Beta Activity of Particulates in Air

Year	Period	# samples (range)	Max*	Min*	Average*	% of MPC [†]
1959	annual	49-52	81.31	0.08	15.76	1.60
1960	Q1	13	2.99	0.24	1.08	0.11
1960	Q2	13	4.22	0.21	1.63	0.16
1960	Q3	14	2.86	0.07	0.85	0.09
1960	Q4	13	1.80	0.04	0.46	0.05
1961	Q1	13-14	1.65	0.00	0.60	0.06
1961	Q2	13-14	8.51	0.18	1.19	0.12
1961	Q3	14	157.00	0.07	20.90	2.10
1961	Q4	13	73.00	16.00	35.00	3.50
1962	Q1/Q2	26-74	90.00	22.00	41.00	4.10
1962	Q3/Q4	26-74	81.00	11.00	30.00	3.00
1963	Q1/Q2	26-181	131.00	27.00	60.00	6.00
1963	Q3/Q4	26-180	69.00	3.00	20.00	2.00
1964	Q1/Q2	26-180	35.00	4.00	13.00	1.30

* Units of 10^{-13} mCi/cc.

[†] Maximum Permissible Concentration (MPC) is taken to be 10^{-10} mCi/cc as recommended in NBS Handbook 69.

The highest percent MPC values for the perimeter and remote monitoring stations for the period were in the first half of 1963 and were reported as 6% and 6.3%, respectively. The news release for that period states, “Although these values are approximately two times greater than the average for the last half of 1962, they are no greater than the average of those measured in other areas of the United States and reported by the U.S. Public Health Radiation Surveillance Network for the period January through May, 1963.”

**Table 2.8-2 Continuous Air Monitoring Data – Remote Stations
Long-Lived Gross Beta Activity of Particulates in Air**

Year	Period	# samples (range)	Max*	Min*	Average*	% of MPC [†]
1959	annual	26-52	100.52	0.14	13.97	1.40
1960	Q1	13	2.73	0.12	1.14	0.11
1960	Q2	10-13	3.11	0.08	1.65	0.17
1960	Q3	11-13	2.39	0.16	0.80	0.08
1960	Q4	12-13	2.66	0.12	0.49	0.05
1961	Q1	13-14	1.18	0.00	0.55	0.06
1961	Q2	13-14	2.22	0.20	0.95	0.10
1961	Q3	14	220.00	0.07	23.60	2.40
1961	Q4	13	88.00	15.00	41.00	4.10
1962	Q1/Q2	26	97.00	20.00	49.00	4.90
1962	Q3/Q4	26	159.00	11.00	36.00	3.60
1963	Q1/Q2	25-26	114.00	35.00	63.00	6.30
1963	Q3/Q4	25-26	91.00	4.00	24.00	2.40
1964	Q1/Q2	25-26	48.00	4.00	17.00	1.70

* Units of 10^{-13} mCi/cc.

[†] Maximum Permissible Concentration (MPC) is taken to be 10^{-10} mCi/cc as recommended in NBS Handbook 69.

Water Monitoring

Liquid wastes originating at ORGDP and the Y-12 Complex were discharged to East Fork Poplar Creek which flows into the Clinch River (ORNL discharged aqueous waste to the Clinch River upstream of ORGDP). River monitoring was performed so that the resulting average concentrations in the Clinch River from all Oak Ridge DOE operations complied with the maximum permissible levels for populations adjacent to DOE (then, AEC) facilities as recommended by the National Committee on Radiation Protection

(NCRP). Radioactive liquid wastes were sampled at a number of locations in the Clinch River, beginning at a point of entry of wastes into the river (mile 20.8) and ending at Center's Ferry near Kingston, Tennessee (mile 4.5). The average concentration of radioactivity at these two points was then calculated. The average concentration of transuranic alpha emitters at mile 20.8 was also calculated. Stream gauging operations were carried on continuously by the U.S. Geological Survey to obtain dilution factors for calculating the probable concentrations of wastes in the river. The average activity in East Fork Poplar Creek was also reported in 1959 and 1960. The results for the five-year period are shown in Table 2.8-3 as percentages of the MPC_w for populations in the neighborhood of a controlled area.

Table 2.8-3 ORGDP Water Monitoring Data

Year	Period	% MPC _w (Clinch River)*		% MPC TRU alpha emitters (Clinch River)	% MPC activity (Poplar Creek)
		Mile 20.8	Mile 4.5		
1959	year	25.4	22.3	0.0300	0.03
1960	Q1	26.9	16.4	0.0020	0.02
1960	Q2	23.2	7.9	0.0010	0.03
1960	Q3	12.6	4.9	0.0010	0.04
1960	Q4	22.0	17.0	0.0004	
1961	Q1	33.0	13.0	0.0007	
1961	Q2	21.0	7.0	0.0005	
1961	Q3	6.3	3.1	0.0030	
1961	Q4	8.8	5.5	0.0001	
1962	Q1/Q2	8.2	6.2	0.0002	
1962	Q3/Q4	6.4	3.9	0.0003	
1963	Q1/Q2	5.6	3.4	0.0002	
1963	Q3/Q4	3.3	4.0	0.0002	
1964	Q1/Q2	3.5	2.0	<0.0010	

**The fraction of the total beta activity comprised by each isotope was determined from analysis of long-lived radionuclides contained in the effluent, and a weighted average maximum permissible concentration for water (MPC_w) for the mixture of radionuclides was calculated on the basis of the isotopic distribution using the MPC values of each isotope as recommended by the NCRP. The average concentration of gross beta activity in the Clinch River was compared to the calculated MPC_w values. The concentration of uranium was compared with the specific MPC_w value for uranium.*

There were no instances of water release above the long-term MPC.

Gamma Measurements

External gamma radiation levels were measured monthly at a number of locations in the Oak Ridge area. These locations included Solway Gate, Y-12 East Portal, Newcombe Road in Oak Ridge, Gallaher Gate, and White Wing Gate. Measurements were taken

with a Gieger-Muller tube at a distance of three feet above ground, with the results tabulated in mR/hr. These results are shown in Table 2.8-4.

The news releases state, “These average levels were the same as average background levels obtained throughout the United States by the U.S. Public Health Service Radiation Surveillance Network, employing similar methods and detection instruments.”

Table 2.8-4 External Gamma Radiation Levels (mR/hr)

Year	Period	Average
1959	year	0.024
1960	Q1	0.017
1960	Q2	0.020
1960	Q3	0.020
1960	Q4	0.020
1961	Q1	0.015
1961	Q2	0.020
1961	Q3	0.019
1961	Q4	0.020
1962	Q1/Q2	0.027
1962	Q3/Q4	0.031
1963	Q1/Q2	0.028
1963	Q3/Q4	0.023
1964	Q1/Q2	0.014

2.8.3 DOE Joint Task Force on Uranium Recycle Materials Processing

A joint task force was assembled by the Department of Energy in 1985 to study past and current practices relating to the processing of uranium recycle materials. From the data reviewed, the task force did not disclose any instance in which the environment, safety, or health of plant workers or the public were jeopardized or compromised. The primary recommendation for all DOE sites from this study was to develop formal, mutually agreeable shipper/receiver specifications on maximum permissible levels of constituents in recycled uranium materials. No specific recommendations were suggested regarding the releases from the Y-12 Complex. This study is documented in DOE/OR-859, *Report of the Joint Task Force on Uranium Recycle Materials Processing*, issued in September 1985.

2.8.4 Oak Ridge Dose Reconstruction Project

An Oak Ridge Dose Reconstruction Project was initiated in 1994 as follow-up to the Oak Ridge Dose Reconstruction Feasibility Study, which recommended a closer examination of past uranium emissions and potential resulting exposures. The initial feasibility study performed screening calculations to identify those operations and materials that warranted detailed investigation in terms of potential off-site exposures to the individuals that have lived in the areas surrounding ORR. At the close of the feasibility study, the Tennessee Department of Health and the Oak Ridge Health Agreement Steering Panel (ORHASP) recommended that a detailed project including dose reconstruction be performed. The results of a portion of this project were documented in the July-1999 Task 6 report titled *Uranium Releases from the Oak Ridge Reservation – A Review of the Quality of Historical Effluent Monitoring Data and a Screening Evaluation of Potential Off-Site Exposures*.¹²

The Task 6 component of the project involved further evaluation of Oak Ridge uranium operations and effluent monitoring records to determine if uranium releases from ORR (including the cumulative effects of releases from all DOE-ORR facilities) likely resulted in off-site doses that warranted further study. The team performed a historical review of air and water release data, including health physics and industrial hygiene reports, stack monitoring data, accident and investigation reports, logbooks, and procedures for the period 1944 through 1988.

Estimates of uranium releases for individual exhaust stacks and building vents were tabulated by the Project Team from original Y-12 Complex documents and included two basic types of release information: (1) reported releases for individual buildings or uranium processes and (2) exhaust stack or indoor air monitoring data and quantities of air exhausted from individual buildings or exhaust stacks. For unmonitored releases or for sampling periods where there was limited data, the Project Team used uranium production rates or release estimates for preceding or subsequent years for which sampling data were available.

For operating periods for which monitoring data were available, the Project Team used uranium concentrations determined from air samples in combination with the amount of air exhausted through stacks and building vents to estimate the quantity of uranium routinely or accidentally released during a particular sampling period.

The Task 6 team concluded that estimates of uranium releases were underestimated by the AEC, DOE, and ORR site contractors. Based on discussion with Y-12 Complex workers, unmonitored release sources were almost exclusively associated with depleted uranium operations and would account for the majority of the differences between the Task 6 and DOE release estimates.¹³ These estimates are shown in Table 2.8-5.

The screening evaluation of potential off-site exposure to waterborne uranium was based on environmental measurements of uranium in local surface waters. Reported annual average uranium concentrations in the Clinch River were used for the Task 6 screening evaluation. These values were based on water samples collected at the confluence of Poplar Creek and the Clinch River for all the years of operation up to 1995.

¹² Buddenbaum et al., *Uranium Releases from the Oak Ridge Reservation- A Review of the Quality of Historical Effluent Monitoring Data and a Screening Evaluation of Potential Off-Site Exposures*, 1999.

¹³ Personal communication between Edward Owings (former Y-12 worker) and the Task 6 team, July 1997.

Table 2.8-5 Y-12 Complex Airborne Uranium Release Estimates

Year	Task 6 Estimate (kg)	DOE Estimate (kg)	Year	Task 6 Estimate (kg)	DOE Estimate (kg)
1944	310	55	1970	300	259
1945	670	102	1971	580	290
1946	390	102	1972	870	222
1947	250	55	1973	410	206
1948	650	0	1974	210	207
1949	650	0	1975	210	209
1950	650	0	1976	210	207
1951	650	0	1977	210	206
1952	650	0	1978	210	205
1953	4000	30	1979	210	206
1954	3800	32	1980	220	218
1955	3800	32	1981	210	207
1956	3000	43	1982	210	207
1957	2300	41	1983	210	208
1958	5700	41	1984	330	329
1959	6200	120	1985	210	210
1960	930	99	1986	210	211
1961	1300	109	1987	150	116
1962	1400	100	1988	150	116
1963	2100	103	1989*		44
1964	2700	170	1990*		21
1965	640	281	1991*		21
1966	920	212	1992*		7
1967	340	212	1993*		3
1968	440	211	1994*		24
1969	250	223	1995*		2
			TOTAL	50,000	6,535

** Values for these years were based on releases reported by DOE. Release estimates for these late years were not independently reconstructed by the Project Team.*

Source: DOE Estimates for years 1944 to 1988 compiled from USDOE 1988; estimates for years 1989 to 1995 were from LMES 1996. Task 6 estimates are rounded to two significant figures.

Effluent monitoring data were also evaluated for quality and consistency with previous DOE historical uranium release reports. The average annual concentration of uranium in the Clinch River for the period 1944 to 1995 was estimated to be 0.015 mgL⁻¹.

Based on the decision guidelines from the ORHASP, the Task 6 team concluded that the Y-12 Complex uranium releases are candidates for further study, but that they are not high-priority candidates.

The Task 7 component of the Oak Ridge Dose Reconstruction effort involved the screening of additional potential materials of concern, including neptunium and technetium. This portion of the effort was documented in the July-1999 Task 7 report.¹⁴

Neptunium

No historical stack monitoring or ambient air monitoring data for Np were identified by the Task 7 team. Therefore, Np sources were estimated based on the total amount of recycled uranium received at the Y-12 Complex from ICPP and SRS for each year from 1953 to 1984.¹⁵ These receipts are shown in Table 2.8-6.

Np concentrations were calculated based on the upper alpha activity of 200,000 dpm g⁻¹ of uranium.¹⁶ Np releases to air from the Y-12 Complex were estimated by calculating a release fraction from the inventory differences for natural uranium reported by Owings.¹⁷ The calculated natural uranium release fraction based on inventory differences was 0.1%. Because the inventory difference value does not distinguish between releases to either air or water, the Project Team relied on its knowledge of uranium processing at the Y-12 Complex to estimate the fraction of the inventory difference that might have been released to air and water. In this analysis, it was assumed that one quarter of the 0.1% inventory difference was released to the air, while three quarters was released to water. The estimated release fraction to air (0.025%) was then multiplied by the Y-12 Complex Np activity inventories to estimate yearly release to air. Similarly, the estimated release fraction to water (0.075%) was multiplied by the Y-12 Complex Np activity inventories to estimate the yearly releases to water. Table 2.8-7 provides the estimated airborne and water releases of Np per year from the Y-12 Complex for the period 1953 to 1995.

Technetium

No airborne effluent information for the Y-12 Complex was located by the Task 7 team. The basis for the estimate of airborne Tc from the plant was, again, the total amount of recycled uranium received from ICPP and SRS between 1953 and 1984 (see Table 2.8-6). The yearly masses of uranium received were multiplied by the estimated Tc concentration in the recycled uranium to arrive at an estimate of the total Tc activity at the Y-12 Complex. Based on information in the ORGDP mass balance document,¹⁸ the team assumed a Tc concentration of 7 ppm in the recycled uranium. The material

¹⁴ Bruce, *Screening-Level Evaluation of Additional Potential Materials of Concern*, 1999.

¹⁵ Egli et al., *The Report of the Joint Task Force on Uranium Recycle Materials Processing*, 1985.

¹⁶ Ibid.

¹⁷ Owings, E., *Historical Review of Accountable Nuclear Materials at the Y-12 Plant*, 1995.

¹⁸ Oak Ridge Gaseous Diffusion Plant, *Draft Mass Balance, ORGDP*, 1978.

Table 2.8-6 Reported Y-12 Complex Receipts of Recycled Uranium

Year	SRP (kg U)	ICPP (kg U)	Total (kg U)
1953	0	101	101
1954	0	217	217
1955	3	828	831
1956	0	744	744
1957	201	797	998
1958	258	898	1,156
1959	270	3,741	4,011
1960	6,395	769	7,164
1961	2,305	0	2,305
1962	2,701	775	3,476
1963	6,461	0	6,461
1964	2,977	771	3,748
1965	3,546	425	3,971
1966	3,467	1,408	4,875
1967	2,604	0	2,604
1968	2,097	394	2,491
1969	4,121	427	4,548
1970	2,045	108	2,153
1971	3,805	1,660	5,465
1972	4,716	415	5,131
1973	5,051	563	5,614
1974	4,599	0	4,599
1975	5,110	1,702	6,812
1976	4,320	195	4,515
1977	4,497	1,333	5,830
1978	2,070	525	2,595
1979	4,591	535	5,126
1980	1,510	0	1,510
1981	4,918	905	5,823
1982	5,728	577	6,305
1983	6,682	1,041	7,723
1984	5,776	2,868	8,644
TOTAL	102,824	24,722	127,546

Note: Historical data from Egli et al., The Report of the Joint Task Force on Uranium Recycle Materials Processing, 1985; does not necessarily agree with the findings of this study as given in Table 3.2-1.

Table 2.8-7 Estimated Y-12 Complex Np Releases

Year	Air Release (μCi)	Water Release (μCi)
1953	2.3	6.8
1954	4.9	15
1955	19	56
1956	17	50
1957	22	67
1958	26	78
1959	90	270
1960	160	480
1961	52	160
1962	78	230
1963	150	440
1964	84	250
1965	89	270
1966	110	330
1967	59	180
1968	56	170
1969	100	310
1970	48	150
1971	120	370
1972	120	350
1973	130	380
1974	100	310
1975	150	460
1976	100	310
1977	130	390
1978	58	180
1979	120	350
1980	34	100
1981	130	390
1982	140	430
1983	170	520
1984	190	580
1985	10	100
1986	10	100
1987	10	100
1988	10	100
1989	10	100
1990	10	100
1991	10	100
1992	10	100
1993	10	100
1994	10	100
1995	10	100
Total (μCi)	2,969.2	9,732.8

balance document states that Paducah personnel estimated government reactor recycled uranium at 7 ppm Tc and that this estimate is consistent with ORGDP data. The mass of Tc received was then calculated using the following equation:

$$Tc (mg) = U (kg) \times Tc \text{ concentration } (mg \text{ kg}^{-1})$$

The mass of Tc received in the recycled uranium in 1953 would then be the following:

$$Tc (mg) = 101 \text{ kg} \times 7 \text{ mg kg}^{-1} = 707 \text{ mg Tc}$$

The activity of Tc received was calculated by multiplying the mass of Tc by the specific activity of Tc ($1.7 \times 10^{-2} \text{ Ci g}^{-1}$):

$$Tc (Ci) = (0.707 \text{ g Tc}) \times (1.7 \times 10^{-2} \text{ Ci g}^{-1}) = 0.012 \text{ Ci}$$

The next step in determining the Tc source term was to define the amount of Tc released to the air. This was accomplished by calculating a release fraction based on the inventory differences for natural uranium at the Y-12 Complex reported by Owings.¹⁹ Inventory difference values were once termed “material unaccounted for” (MUF). The calculated natural uranium release fraction based on inventory differences was 0.1%. This value was multiplied by the Y-12 Complex Tc activity inventories to yield conservative annual airborne release estimates. The results of this analysis are presented in Table 2.8-8.

No measurements of Tc concentrations in liquid effluent from the plant prior to the late 1980s were identified by the Project Team. Beginning in 1991, concentrations of Tc were measured monthly in East Fork Poplar Creek at the junction of Bear Creek and Scarboro Roads. The concentrations ranged from less than background to 160 pCiL^{-1} . Individual sample results were not located.

In addition to routine monitoring, two special studies also measured Tc concentrations in surface waters around the ORR. The potential source of these Tc concentrations was not limited to the Y-12 Complex.

- The *Instream Contaminant Study* – the only surface water sample analyzed for Tc as part of this study was located in Watts Bar Reservoir at Clinch River Mile 6.8. The concentration of Tc in this sample was 0.73 pCiL^{-1} .²⁰
- The *Clinch River Remedial Investigation* – Tc concentrations in the Clinch River ranged from less than the limit of detection to 23 pCiL^{-1} . The Poplar Creek concentrations ranged from less than the limit of detection to 32 pCiL^{-1} .²¹

¹⁹ Owings, *Historical Review of Accountable Nuclear Materials at the Y-12 Plant*, 1995.

²⁰ Tennessee Valley Authority, *Instream Contaminant Study*, 1985.

²¹ Cook et al., *Phase I Data Summary Report for the Clinch River Remedial Investigation*, 1992.

Table 2.8-8 Estimated Tc Releases from the Y-12 Complex

Year	Estimated Tc Release (Ci)
1953	1.2×10^{-5}
1954	2.6×10^{-5}
1955	9.9×10^{-5}
1956	8.9×10^{-5}
1957	1.2×10^{-4}
1958	1.4×10^{-4}
1959	4.8×10^{-4}
1960	8.5×10^{-4}
1961	2.7×10^{-4}
1962	4.1×10^{-4}
1963	7.7×10^{-4}
1964	4.5×10^{-4}
1965	4.7×10^{-4}
1966	5.8×10^{-4}
1967	3.1×10^{-4}
1968	3.0×10^{-4}
1969	5.4×10^{-4}
1970	2.6×10^{-4}
1971	6.5×10^{-4}
1972	6.1×10^{-4}
1973	6.7×10^{-4}
1974	5.5×10^{-4}
1975	8.2×10^{-4}
1976	5.4×10^{-4}
1977	6.9×10^{-4}
1978	3.1×10^{-4}
1979	6.1×10^{-4}
1980	1.8×10^{-4}
1981	6.9×10^{-4}
1982	7.5×10^{-4}
1983	9.2×10^{-4}
1984 through 1995	1.0×10^{-3} each year